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Abstract

Recent Development of Beryllium in Japan

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This paper describes the outline of the recent research and development of beryllium metal and beryllium oxide for nuclear reactor materials with special reference to fuel matrix or fuel cannings and other reactor components for the reactor with high temperature and high burn up.

1. Trends of Research and Development:

Research and development of beryllium-base materials for nuclear reactor use is believed to be important from the point of view of nuclear fuel situation of this country, that is, it is better to avoid to use enriched uranium fuel as possible as it can be. This research and development also must be necessary for development of nuclear reactor with high temperature and high burn up.

From another point of view, the demands for beryllium metal is not so big, and so its price is rather high. Therefore, great interest has been indicated to the utilization of beryllium oxide which is cheeper than the metal, as nuclear reactor material.

Research and development of the utilization of beryllium oxide should be continued more and more as well as the efforts to produce cheeper beryllium metal.

2. Research of Fubrication of Beryllium Pipe by Means of Extrusion Method:

Beryllium metal has been considered as one of the best materials for fuel cannings for the reactor of natural uranium - Co_2 cooled type.

As the first step of the preparation of fuel cannings, fabrication of bellium pipe by extrusion was investigated and the high temperature properties were also examined by high temperature hardness testing.

Electrolytically prepared beryllium (Boron content less than 2 ppm) was melted in vacuum to remove chlorine that was contained in original flaky beryllium with the amount of 2000 ppm approximately. By this treatment, it was reduced to about 50 ppm.

Beryllium billet for extruding prepared by the process of vacuum melting-powdering-vacuum hotpressing-machining was fully sheathed by mild steel. After preheating the billet in fused NaCl - 36 mol % $BaCl_2$ to 1150°C, it was placed into billet container and then extruded with the extrusion ratio of 1/4.

The effect of purity of beryllium on the extrusion characteristics was checked, but remarkable deference was not observed.

Tab. 1 Chemical Analysis of Beryllium Flake Used in the Experiment (ppm)

	Fe	Al	Cu	Ni
Higher grade	350	90	70	140
Lower grade	2300	300	280	300

Tab. 2 Tensile Properties of Extruded Beryllium (Longitudinal direction, as extruded)

Tensile Strength	Elongation
44 kg/mm^2 (60,000 lb/in^2)	12 %

Longitudinal tensile properties as extruded state was shown in tab. 2. Ductility in this direction was sufficient considerably.

As the results of high temperature hardness measurement, no remarkable plateau in temperature-hardness ourve was appeared.

R. Thomson et al - J. Lesscomm Metals, 3 (1961), 170 - reported the appearance of plateau in extruded bar of thermally reduced pebble beryllium, and Kaufmann - Trans. ASM, 42 (1950), 785 - showed the decrease of elongation in the temperature range between 400 to 800°C in the same material.

Supposedly our extruded and anneal beryllium may show no decrease of elongation in this temperature range. This problem is now under investigation as well as transverse ductility problem and these details will be obtained in the near future.

3. Recent Advances in Beryllium Oxide:

Most efforts of research has been directed to develop fuel matrix material and reflector or moderator material in beryllium oxide field.

3.1. Highly Sinterable Beryllium Oxide Powder:

High purity and high sinterable beryllium oxide powder called CF-grade BeO by the process of some combination of recrystallization and thermal decomposition of beryllium sulphate and fabrication of shapes of this powder have been investigated.

Tab. 3 Relationship between Sintering Temperature and Density of CF-grade Beryllium Oxide Powder and Conventional Powder

Powder Grade	Sintering Temperature	Density
CF	1400°C	greater than 2.90 g/cm^3
CF	1500°C	greater than 2.95 g/cm^3
CF	1600°C	greater than 2.98 g/cm^3
Conventional	1600°C	2.80 ~ 2.90

This powder was more sinterable than conventional powder as shown in tab. 3.

Boron and Lithium content of this powder is below 0.5 ppm, and below 0.5 ppm respectively and so that chemical and nuclear purity were both very high. Therefore, this is sufficient for nuclear use. Particle size of this powder was estimated to be finer than conventional powder (0.1 ~ 0.2 μ) by electron microscope and BET method.

3.2. BeO - Be Shapes:

Mechanical strength as well as the superiority of nuclear and thermal properties is important to utilize beryllium oxide shapes as nuclear reactor materials.

This study was intended to develop high strength beryllium oxide shapes.

High purity beryllium oxide powder with 0.1 to 5 wt % of metallic beryllium powder was hot pressed in vacuum at the temperatures of 1200 to 1500°C, and the relationships among sintering condition, composition, and characteristics of sintered body was investigated and BeO-Be shapes with bending strength of 2000 to 3000 kg/cm^2 (28,000 to 43,000 lb/in^2) and density of 99% up of theoretical or more was easily obtained.

This value of bending strength of BeO-Be shapes was higher than BeO mono-component shapes of the same density. Besides, thermal expansion coefficient and thermal conductivity were approximately equal to those of BeO mono-component shapes.

This BeO-Be can be considered as reflector or moderator material and fuel matrix material (BeO-Be-UO_2 & BeO-Be-ThO_2U_2) especially for high temperature reactors.